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Abstract

CALCULATION OF THERMAL NEUTRONS UTILIZATION COEFFICIENT
IN CELLS WITH COMPLEX FUEL ELEMENTS

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Core lattices of many reactors contain instead of individual cylindrical fuel elements bunches of elements-channels surrounded by a layer of moderating material, which is longer than the free path of thermal neutrons. Thus ice-breaker "Lenin" reactors and physical and technical research reactor RFT have the canal structure. If the channel is split into a small number of cores it might lead to a considerable error in the calculations of thermal neutrons utilization coefficient. If one takes into account the channel structure and the change of its composition in the process of burning-out one has to use finite difference methods with several scores of nodes along the cell radius. Application of artificial methods, such as integral method of Amuajal and Benua is practically impossible. Solution of kinetic equation by expanding into spherical harmonics does not ensure the required accuracy if there is a thin absorber-type control rod (for example, a regulator) inside the channel. The Monte-Carlo method is probably possible only for the channels with low symmetry (such as first atomic power station channels). The methods of characteristics (Vladimirov) and of S_n (Carlson) seem to be more convenient for the symmetrical cells.

The computation time is greatly increased by combining the most precise possible account of thermalization and kinetic effects in one programme for the geometrically complex cells. It is rational to analyse the thermalization separately, in diffusion approximation and to use the obtained spectra for averaging cross-sections. Thus the problem is reduced to a solution of a single-velocity kinetic equation with cross-sections which are piece by piece continuous as function of co-ordinates.

The paper gives a description of a diagram and of a programme of this type of calculations. The estimation of thermalization effects is based upon assumptions of Maxwellian (with variable temperature) nature of thermal neutron spectrum in the moderator and of absence of energy exchange between neutrons and fuel nuclei. The kinetic equation is solved by means of a somewhat modified method of characteristics. The number of characteristics may be less than that of the nodes along the radius. The data of P_1 or P_3 solutions are used as the first approximation. Iterations are done only externally, over the source. The diagram is approved of by comparing the results with the known data, including the experimental data for simple uranium-water lattices. In this case the coincidence turns out to be better than satisfactory.

In case of multi-layer cells the analyses of flux distribution and of thermal neutrons utilization coefficient values (obtained by P_1 , P_3 and S_n methods as well) make it possible to arrive at a conclusion as to the necessary detailed structure of cell geometry and to the required accuracy of equation solution.